Docket No. 50-362

MAY 18 1984

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Gentlemen:

Subject: Issuance of Amendment No. 11 to Facility Operating License NPF-15
San Onofre Nuclear Generating Station, Unit 3

The Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 11 to Facility Operating License NPF-15 for the San Onofre Nuclear Generating Station, Unit 3, located in San Diego County, California. The amendment modifies the Technical Specifications to (1) delete a description of the specific methodology used to calculate the minimum Departure from Nucleate Boiling Ratio (DNBR) trip setpoint from the safety system settings and (2) add a description of the specific methodology used to calculate the minimum DNBR trip setpoint from the safety system settings. This amendment was requested by your letters of January 25, July 14 and September 23, 1983.

A copy of the Safety Evaluation supporting this amendment is also enclosed.

Sincerely,

George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

Enclosures:
1. Amendment No. 11 to NPF-15
2. Safety Evaluation

cc w/enclosures:
See next page

HR DL:LB#3	DL:LB#3	OELD	DL:LB#3
HRood/yt	JLee	LChandler	GKnighton
4/30/84	4/ /84	5/9/84	5/16/84

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SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-362

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 11
License No. NPF-15

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the license for San Onofre Nuclear Generating Station, Unit 3 (the facility) filed by the Southern California Edison Company on behalf of itself and San Diego Gas and Electric Company, The City of Riverside and The City of Anaheim, California (licensees) dated January 25, 1983, as supplemented by letters dated July 14 and September 23, 1983 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public;

E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. NPF-15 hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 11, are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: MAY 18 1984

DL:LB#3	DL:LB#3	OELD	DL
HRood/yt	JLee	LChandler	GWKnighton
4/27/84	5/1/84	5/9/84	5/16/84

SAFETY EVALUATION

AMENDMENT NO. 11 TO NPF-15

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 3

DOCKET NO. 50-362

Introduction and Summary

By letters dated January 25, July 14, and September 23, 1983, the licensees (Southern California Edison Company, San Diego Gas and Electric Company, the City of Anaheim, California, and the City of Riverside, California) requested that an amendment be issued to Facility Operating License NPF-11 for operation of the San Onofre Nuclear Generating Station, Unit 3. The change results in a reduction of the Departure from Nucleate Boiling Ratio (DNBR) rod bow penalty factors of values supported by the NRC-approved Combustion Engineering Topical Report CENPD-225P, "Fuel and Poison Rod Bowing."

Evaluation

Specifically the amendments requested by the licensees make the following changes:

(1) Note 5 in Table 2.2-1 of Technical Specifications 2.2-1 is changed by the deletion of a description of the specific methodology used to calculate the minimum Depature from Nucleate Boiling Ratio (DNBR) trip setpoint from the safety system settings.

(2) Section B 2.2.1 of the Technical Specifications is modified by the addition of a description of the specific methodology used to calculate the minimum DNBR trip setpoint from the safety system settings. The methodology differs from that deleted from Note 5 of Table 2.2-1 in that it includes metholology for incorporation of rod bow penalty factors into the Core Operating Limit Supervisory System (COLSS) and Core Protection Calculator (CPC) calculations of DNBR.

(3) The ACTION statement of Technical Specification 3/4.2.4 is changed by requiring the plant operators to "restore" the DNBR to within acceptable limits if it goes outside such limits. The present wording requires the operators to "reduce" the DNBR to within acceptable limits if it goes outside such limits.

(4) Technical Specifications 4.2.4.4 and B 3/4.2.4 are changed to incorporate revised, burnup-dependent DNBR rod bow penalty factors. The revised factors are based on Combustion Engineering Topical Report CENPD-225P.

Items (1) and (3) above are editorial in nature. Items (1) and part of (2) merely change the location of the description of the methodology used to calculate the minumum DNBR from Note 5 of Table 2.2-1 to Section B.2.2.1. Item (3) changes the word "reduce" to "restore", because in some cases, the DNBR may have to be increased, rather than reduced, to restore it to within acceptable limits. Thus, changes (1) and (3) involve no significant hazards consideration.

Items (2) and (4) above incorporate the revised, burnup-dependent rod bow penalty factors. As is discussed in CENPD-225P, the revised penalty factors are based on experimental data taken from 16 x 16 fuel element assemblies, such as those used in San Onofre 2. The previous rod bow penalty factors were based on a conservative extrapolation of data from 14 x 14 fuel element assemblies. Thus, the revised penalty factors are based on data which is more directly applicable to San Onofre 2. Both sets of penalty factors were selected to provide a 95% probability with 95% confidence that DNB will not occur on a fuel rod having the minimum DNBR during steady-state operation and anticipated operational occurrences. The revised rod bow penalty factors are in agreement with those given in the Combustion Engineering Topical Report, CENPD-225P and its supplements, "Fuel and Poison Rod Bowing." This report has been reviewed and approved by the NRC staff in a letter from C. O. Thomas (NRC) to A. E. Scherer (CE) dated February 15, 1983.

Thus, while the use of the revised rod bow penalty factors may, under some operating conditions, reduce a safety margin, the results of the change are clearly within all acceptable safety criteria. In particular, the revised penalty factors provided the same 95/95 percent probability and confidence that DNBR will not be exceeded. Therefore, this amendment is essentially the same as item (vi) of the examples of actions involving no significant hazards consideration given in 48 FR 14870 and is acceptable.

Contact With State Official

By copy of a letter dated December 15, 1983 to the licensees, the NRC staff advised the Chief of the Radiological Health Branch, State Department of Health Services, State of California, of its proposed determination of no significant hazards consideration. No comments were received.

Environmental Consideration

We have determined that these amendments do not authorize a change in effluent types of total amount nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that these amendments involve action which is insignificant from the standpoint of environmental impact and pursuant 10 CFR Section 51.5(d) (4), that an environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

Conclusion

Based upon our evaluation of the proposed changes to the San Onofre, Unit 3 Technical Specifications, we have concluded that: there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public. We, therefore, conclude that the proposed changes are acceptable.

Dated: MAY 18 1984

HR	
DL:LB#3	DL:LB#3
HRood/yt	GWKnighton
4/30/84	5/16/84

MAY 18 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 11

FACILITY OPERATING LICENSE NO. NPF-15

DOCKET NO. 50-362

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Also to be replaced are the following overleaf pages to the amended pages.

Amendment Pages	Overleaf Pages
2-4	2-3
B 2-6	B 2-5
B 2-7	-
B 2-8	-
3/4 2-6	3/4 2-5
B 3/4 2-4	B 3/4 2-3

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High -		
Four Reactor Coolant Pumps Operating	$\leq$ 110.0% of RATED THERMAL POWER	$\leq$ 111.3% of RATED THERMAL POWER
3. Logarithmic Power Level - High (1)	$\leq$ 0.89% of RATED THERMAL POWER	$\leq$ 0.96% of RATED THERMAL POWER
4. Pressurizer Pressure - High	$\leq$ 2382 psia	$\leq$ 2389 psia
5. Pressurizer Pressure - Low (2)	$\geq$ 1806 psia	$\geq$ 1763 psia
6. Containment Pressure - High	$\leq$ 2.95 psig	$\leq$ 3.14 psig
7. Steam Generator Pressure - Low (3)	$\geq$ 729 psia	$\geq$ 711 psia
8. Steam Generator Level - Low	$\geq$ 25% (4)	$\geq$ 24.23% (4)
9. Local Power Density - High (5)	$\leq$ 19.95 kw/ft	$\leq$ 19.95 kw/ft
10. DNBR - Low	$\geq$ 1.20 (5)	$\geq$ 1.20 (5)
11. Reactor Coolant Flow - Low		
a) DN Rate	$\leq$ 0.22 psid/sec (6)(8)	$\leq$ 0.231 psid/sec (6)(8)
b) Floor	$\geq$ 13.2 psid (6)(8)	$\geq$ 12.1 psid (6)(8)
c) Step	$\leq$ 6.82 psid (6)(8)	$\leq$ 7.231 psid (6)(8)
12. Steam Generator Level - High	$\leq$ 90% (4)	$\leq$ 90.74% (4)
13. Seismic - High	$\leq$ 0.48/0.60 (7)	$\leq$ 0.48/0.60 (7)
14. Loss of Load	Turbine stop valve closed	Turbine stop valve closed

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATION

(1) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(2) Value may be decreased manually, to a minimum value of 300 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(3) Value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and low level instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. Trip may be manually bypassed below 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 10^{-4}% of RATED THERMAL POWER. The approved DNBR limit accounting for use of HID-2 grid is 1.2.0. A DNBR trip setpoint of 1.19 is allowed provided that the difference is compensated by an increase in the addressable constants BERRI for CPC and EPOL2 for COLSS.

(6) DN RATE is the maximum decrease rate of the trip setpoint.

FLOOR is the minimum value of the trip setpoint.

STEP is the amount by which the trip setpoint is below the input signal unless limited by DN Rate or Floor.

(7) Acceleration, horizontal/vertical, g.

(8) Setpoint may be altered to disable trip function during testing pursuant to Specification 3.10.3.

Local Power Density-High (Continued)

The local power density (LPD), the trip variable, calculated by the CPC incorporates uncertainties and dynamic compensation routines. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core peak LPD is sufficiently less than the fuel design limit such that the increase in actual core peak LPD after the trip will not result in a violation of the peak LPD Safety Limit. CPC uncertainties related to peak LPD are the same types used for DNBR calculation. Dynamic compensation for peak LPD is provided for the effects of core fuel centerline temperature delays (relative to changes in power density), sensor time delays, and protection system equipment time delays.

DNBR-Low

The DNBR - Low trip is provided to prevent the DNBR in the limiting coolant channel in the core from exceeding the fuel design limit in the event of anticipated operational occurrences. The DNBR - Low trip incorporates a low pressurizer pressure floor of 1825 psia. At this pressure a DNBR - Low trip will automatically occur. The DNBR is calculated in the CPC utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore neutron flux monitoring system;

b. Reactor Coolant System pressure from pressurizer pressure measurement;

c. Differential temperature (Delta T) power from reactor coolant temperature and coolant flow measurements;

d. Radial peaking factors from the position measurement for the CEAs;

e. Reactor coolant mass flow rate from reactor coolant pump speed;

f. Core inlet temperature from reactor coolant cold leg temperature measurements.

The DNBR, the trip variable calculated by the CPC incorporates various uncertainties and dynamic compensation routines to assure a trip is initiated prior to violation of fuel design limits. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core DNBR is sufficiently greater than 1.20 such that the decrease in actual core

DNBR-Low (Continued)

DNBR after the trip will not result in a violation of the DNBR Safety Limit. CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modelling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays.

The DNBR algorithm used in the CPC is valid only within the limits indicated below and operation outside of these limits will result in a CPC initiated trip.

a.	RCS Cold Leg Temperature-Low	$\geq$ 495°F
b.	RCS Cold Leg Temperature-High	$\leq$ 580°F
c.	Axial Shape Index-Positive	< +0.5
d.	Axial Shape Index-Negative	$\geq$ -0.5
e.	Pressurizer Pressure-Low	$\geq$ 1825 psia
f.	Pressurizer Pressure-High	$\leq$ 2375 psia
g.	Integrated Radial Peaking Factor-Low	$\geq$ 1.28
h.	Integrated Radial Peaking Factor-High	$\leq$ 4.28
i.	Quality Margin-Low	< 0

The DNBR Trip setpoint in CPC and COLSS is 1.19. The values of the penalty factors BERR1 (CPC) and EPOL2 (COLSS) may be adjusted to implement requirements for tripping at other values of DNBR. The following formula is used to adjust the CPC addressable constant BERR1:

$$BERR1_{new} = BERR1_{old}\ [1 + \Delta DNBR(\%) * \left|\frac{d(\%\ POL)}{d\ (\%\ DNBR)}\right| * 0.01]$$

where:

$BERR1_{new}$ = new required value of BERR1,

$BERR1_{old}$ = present implemented value of BERR1,

$\Delta DNBR(\%)$ = percent increase in DNBR trip setpoint requirement,

d(% POL)/d(% DNBR) = The absolute value of the most adverse derivative of percent POL with respect to percent DNBR as reported in CEN-184(S)-P.

Similarly, for the COLSS addressable constant EPOL2:

$$EPOL2_{new} = (1 + \Delta DNBR(\%) * \left|\frac{d(\%\ POL)}{d\ (\%\ DNBR)}\right| * 0.01) * (1 + EPOL2_{old}) - 1.0$$

where:

$EPOL2_{new}$ = new required value of EPOL2,

$EPOL2_{old}$ = present implemented value of EPOL2,

and the other terms are as previously defined.

DNBR-Low (Continued)

This illustrates the methodology used for conversion of any DNBR penalty into a format that is useable and addressable in both CPC and COLSS. The addressable constants BERR1 and EPOL2 are also used to accommodate the DNBR rod bow penalties listed in Technical Specification 4.2.4.4

Reactor Coolant Flow - Low

The Reactor Coolant Flow - Low trip provides protection against a reactor coolant pump sheared shaft event and a two pump opposite loop flow coastdown event. A trip is initiated when the pressure differential across the primary side of either steam generator goes below a variable setpoint. This variable setpoint stays a set amount below the pressure differential unless limited by a set maximum decrease rate or a set minimum value. The specified setpoint ensures that a reactor trip occurs to prevent violation of local power density or DNBR safety limits under the stated conditions.

Seismic - High

The Seismic - High trip is provided to trip the reactor in the event of an earthquake which exceeds 60% of the Safe Shutdown Earthquake level. This trip's setpoint does not correspond to a safety limit and no credit was taken in the accident analyses for operation of this trip.

Loss of Load

The Loss of Load trip is provided to trip the reactor when the turbine is tripped above a predetermined power level. This trip's setpoint does not correspond to a Safety Limit and no credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting enhances the overall reliability of the Reactor Protection System.

Steam Generator Level-High

The Steam Generator Level-High trip is provided to protect the turbine from excessive moisture carry over. Since the turbine is automatically tripped when the reactor is tripped, this trip provides a reliable means for providing protection to the turbine from excessive moisture carry over. This trip's setpoint does not correspond to a Safety Limit and no credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting enhances the overall reliability of the Reactor Protection System.

2.2.2 CPC ADDRESSABLE CONSTANTS

The Core Protection Calculator (CPC) addressable constants are provided to allow calibration of the CPC system to more accurate indications such as calorimetric measurements for power level and RCS flow rate and incore detector signals for axial flux shape, radial peaking factors and CEA deviation penalties. Other CPC addressable constants allow penalization of the calculated DNBR and LPD values based on measurement uncertainties or inoperable equipment. Administrative controls on changes and periodic checking of addressable constant values (see also Technical Specifications 3.3.1 and 6.8.1) ensure that inadvertent misloading of addressable constants into the CPC's is unlikely.

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by operating within the region of acceptable operation of Figure 3.2-1 or 3.2-2, as applicable.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With operation outside of the region of acceptable operation, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR; or (2) when the COLSS is not being used, any OPERABLE Low DNBR channel exceeding the DNBR limit, within 15 minutes initiate corrective action to restore the DNBR to within the limits and either:

a. Restore the DNBR to within its limits within one hour, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR, as indicated on all OPERABLE DNBR channels, is within the limit shown on Figure 3.2-2.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.4.4 The DNBR penalty factors included in the COLSS and CPC DNBR calculations shall be verified at least once per 31 EFPDs to be greater than or equal to the values listed below. This verification will be made on the basis of the BERR1 addressable constant for the CPC and the EPOL2 addressable constant for the COLSS.

Burnup $\frac{GWD}{MTU}$	DNBR Penalty (%)
0-10	0.5
10-20	1.0
20-30	2.0
30-40	3.5
40-50	5.5

AZIMUTHAL POWER TILT - T_q (Continued)

T_q is the peak fractional tilt amplitude at the core periphery

g is the radial normalizing factor

θ is the azimuthal core location

θ_o is the azimuthal core location of maximum tilt

P_{tilt}/P_{untilt} is the ratio of the power at a core location in the presence of a tilt to the power at that location with no tilt.

3/4.2.4 DNBR MARGIN

The limitation on DNBR as a function of AXIAL SHAPE INDEX represents a conservative envelope of operating conditions consistent with the safety analysis assumptions and which have been analytically demonstrated adequate to maintain an acceptable minimum DNBR throughout all anticipated operational occurrences, of which the loss of flow transient is the most limiting. Operation of the core with a DNBR at or above this limit provides assurance that an acceptable minimum DNBR will be maintained in the event of a loss of flow transient.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the DNBR channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the DNBR does not violate its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core operating limit corresponding to the allowable minimum DNBR. Reactor operation at or below this calculated power level assures that the limits of Figure 3.2-1 are not violated. The COLSS calculation of core power operating limit based on the minimum DNBR limit includes appropriate penalty factors which provide, with a 95/95 probability/confidence level, that the core power limit calculated by COLSS (based on the minumum DNBR limit) is conservative with respect to the actual core power limit. These penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering design factors, state parameter measurement, software algorithm modelling, computer processing, rod bow and core power measurement.

Parameters required to maintain the margin to DNB and total core power are also monitored by the CPCs. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-2 can be maintained by utilizing a predetermined DNBR as a function of AXIAL SHAPE INDEX and by monitoring the CPC trip channels. The above listed uncertainty penalty factors plus those associated with startup test acceptance criteria are also included in the CPC's which assume a minimum core power of 20% of RATED THERMAL POWER. The 20% Rated Thermal Power threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings.

BASES

The DNBR penalty factors listed in Section 4.2.4.4 are penalties used to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher average burnup will experience a greater magnitude of rod bow. Conversely, lower burnup assemblies will experience less rod bow. The penalty for each batch required to compensate for rod bow is determined from a batch's maximum average assembly burnup applied to the batch's maximum integrated planar-radial power peak. A single net penalty for COLSS and CPC is then determined from the penalties associated with each batch, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

3/4.2.5 RCS FLOW RATE

This specification is provided to ensure that the actual RCS total flow rate is maintained at or above the minimum value used in the LOCA safety analyses.

3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

This specification is provided to ensure that the actual value of reactor coolant cold leg temperature is maintained within the range of values used in the safety analyses.

2.4.2.7 AXIAL SHAPE INDEX

This specification is provided to ensure that the actual value of AXIAL SHAPE INDEX is maintained within the range of values used in the safety analyses.

3/4.2.8 PRESSURIZER PRESSURE

This specification is provided to ensure that the actual value of pressurizer pressure is maintained within the range of values used in the safety analyses.

Document Name:
SONG AMENDMENT ~~10~~ 11

Requestor's ID:
PEGGY

Author's Name:
HRood/yt

Document Comments:
Issuance of Amendment No. ~~10~~ 11 to FAcility Operating License

May 18, 1984

ISSUANCE OF AMENDMENT NO. 11 TO FACILITY OPERATING LICENSE NPF-15
SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 3

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Docket File 50-362
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